SITUATION OF THE TECHNOLOGICAL IRRADIATION REACTORS
A PROGRESS REPORT ON THE JULES HOROWITZ REACTOR PROJECT

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ABSTRACT

Since the shutdown of the SILOE reactor in 1997, the OSIRIS reactor has ensured all the needs regarding technological irradiation at the CEA, including those of its industrial partners and customers. Within this context, it has undergone regular renovation allowing it to maintain a high level of safety. This situation has led to release the planning of the Jules Horowitz Reactor which is now scheduled to continue up to the year 2010, thereby allowing the project with researchers to take advantage of this prolonged interval in order to optimize reactor performances and pursue studies within the field of possible materials in research reactors and irradiation devices. One of the critical points that limit performances and thus the flux levels of the JHR concern the maximum acceptable temperature on the aluminum cladding. We have formulated the hypothesis that the 150° C temperature should not be reached in any situation whatsoever (except in transients), but this limit must be determined with accuracy. Therefore tests concerning this problem have been carried out. Furthermore, the UMo_7 (8.0 g/cc) fuel was selected as a reference fuel in June 2000 with the U_3Si_2 fuel now being regarded as an alternative or " fallback " solution while awaiting final qualification of the UMo.

1 The French Context

Since the shutdown of the SILOE reactor in 1997, the CEA now has only one technological irradiation reactor (MTR) in operation, the OSIRIS reactor. It has undergone regular renovation allowing it to continue satisfying the most stringent demands in terms of safety. The replacement in 1996 of the aluminum housing (i.e. the component maintaining all fuel elements in place in the core) has been considered to be a major operation and thus a determining factor in prolonging the reactor's lifetime. In 2000, specific attention has been concentrated on the renovation of the liner of the water channels. With the replacement of the Zircaloy tank, planned for 2001, it can safely be stated that all sensitive components will have been replaced by this time.
The safety report, published in November of 1999 concluded that this reactor, in operation since 1966, could still continue to operate on a minimum basis for another 10 years. (The next Safety Decade Report) without any major modification.
The irradiation devices for which it was possible to guarantee the continuity of the programs have been transferred successfully from SILOE to OSIRIS.
The ORPHEE reactor is essentially dedicated to fundamental research. This reactor nevertheless ensures regular production of silicon doping and provides neutronographic services for industry, particularly in the field of space for "ARIANE ESPACE". It supplies radioisotopes for medicine.
The critical assembly (Masurca, Eole, Minerve) is used for specific programs.
The Cabri and Phébus reactors are regularly used for safety experiments.

CEA REACTORS

	TYPE	PRESENT PROGRAMMES
OSIRIS	MTR	-Increase of the reactor lifetime of PWRs -Increase of the burnup of PWR fuels -actinide transmutation studies -MOX fuel qualification
ORPHEE	Fundamental research	-Study of the condensed matter
MASURCA EOLE MINERVE ISIS	Critical assembly Critical assembly Critical assembly Critical assembly	- Study of the hybrid demonstrator configuration - Study of MOX configurations - Study of core configurations - Study of the OSIRIS configurations, dosimetry
CABRI PHEBUS	Safety experiments Safety experiments	- Reactivity injection tests (RIA) - Study of severe accidents

Within this context, and more specifically in order to take advantage of the favorable situation of the OSIRIS reactor regarding safety as much as the realization of experimental programs, the decision was made to release the time schedule of the JHR Project for a commissioning that has now been set at about 2010 rather than the initial date for the year 2006, giving added time for the project performance optimisation and studies on critical aspects of the project, as described below.

2 Progress in the studies on the Jules Horowitz Reactor (JHR)

2.1 Recall of the major landmark stages of the project

PROGRESSION OF THE PROJECT	MAJOR DATES
• Choice of major options - pool reactor, 100 MW -upward flow, pressurized circuit -max speed :15 m/s -aluminum monoblock tank-housing -circular plaque elements - 2 separate buildings . reactor building, . the annex building, housing the hot cells, pools	June 99
• Site file - qualification of the site - transfer of the file to the safety authority	March 2000
• The Safety Option File (DOS) - transfer of the file to the safety authority	June 2000
• launching of the definition studies - CEA-TA contract • preliminary safety report • pouring of the first concrete • the reactor goes critical • commissioning	March 2001 2003 2005 2009 2010

2.2 Recall of the main characteristics

	Type	Pressurized tank in pool
	Power	100 MW
	Height	0.80 m
	Equivalent diameter	0.51 m
Core	Specific power	600 KW/l (300-750 kW/l range)
	Number of fuel elements	37
	Reflector	H_2O-Be
	Flow direction	Upward
	Primary flowrate	1.2 m^3s^{-1}
	Water circulation speed	15.8 $m.s^{-1}$
	Inlet temperature	30°C
Maximum neutronic flux	Thermal flux	7.10^{14} $n.cm^{-2}s^{-1}$
	Fast flux	8.10^{14} $n.cm^{-2}s^{-1}$
	Composition	UMo
	Density	8.0 g $U.cm^{-3}$
Fuel	Enrichment	19.75 %

2.3 Neutronic and Thermal-hydraulic Studies

Before launching the definition studies, the objective of the core studies consists in verifying that the selected options following the preliminary studies will allow researchers to reach the required performances, particularly the flux levels on the materials and fuels irradiated in the devices in the core or outside.

MINIMAL DEMANDS CONCERNING FLUX LEVELS ON THE SAMPLES

Configuration (power/core litre) (KW/l)	Core		Reflector	
	thermal flux $10^{14}n/cm^2/s$	Fast flux $10^{14}n/cm^2/s$	thermal flux $10^{14}n/cm^2/s$	Fast flux $10^{14}n/cm^2/s$
300	2.5	2.5	0.2	2
600 (reference configuration)	2.5	6	4	0.5
750	6	8	2	0.5

thermal flux : ≤ 0.625 ev
rapid flux : ≥ 0.907 Mev

The need to have a whole set of validated codes in order to justify the accuracy of the results obtained, particularly for the safety authority, has led to modifying the numerical schemes of the existing codes in order to take into account the specificities of the Jules Horowitz Reactor (its geometry, temperature, pressure,...).
Moreover, an experimental qualification program is to be carried out on the critical assemblies and test loops.

2.3.1 Thermal-hydraulic qualification tests (program being in progress for a start-up in 2001 at CEA/GRENOBLE)

The parameters of the JHR operation extend the scope of the parameters already evaluated.
A given number of proven models, already available, are potentially applicable for the JHR.
The critical flux is a point which must still be examined and which could become dimensioning if the accuracy of the model is not improved (about 30% at the present time).
We must have a maximum of the measured variables in order to move forward and improve upon what has already been accomplished.
At the present time, no test results exist on the market that can simultaneously cover the JHR conditions (surfacic flux, speed, and water channel size). We shall therefore take the existing correlations (the most adequate) and adjust their coefficients.

A review of the correlations :

Models taken from documentation as well as those that have been established at the CEA have been reviewed for the purpose of determining the pertinence of their use in the case of the JHR.

In order to accomplish this, the tests carried out at the CEA have been re-calculated essentially the CASIMIR, SUPER-BOB tests for which the variation range of the parameters was:

-inlet temperature between 25 and 85 degrees
-pressure between 1 and 11 bars,
-heating length between 60 and 90 cm,
-flux between 200 and 700 W/cm2
-water channel between 1.8 and 3.6 mm.

Here is a review of the list of correlations :

Simple liquid phase :

- Friction (isothermic and anisothermic component) (isothermic : Hägen-Poiseuille, Shah and Bhatti, Blasius, Prandtl-Karman-Nikuradsee, Techo and al., Fajeau and al, Nikuradse, McAdams, Filonenko, Drew and al, Colebrook) (non-isothermic : Costa, Lavigne, Sieder and Tate, Lafay, Ricque and Siboul, Fajeau ...).

- Thermal exchange (Dittus and Boelter, Colburn, Sieder and Tate, Petukhov, Kirillov and Popov, Shah and Bhatti, Gnielinski).

Double-phase :

- Friction (the two contributions: simple-phase + diphasic corrector) (Lockart, Martinelli, Nelson, Thom, Baroczy, Chisollm, HTFS, Friedel, Reddy and al., Whalley, Hetsroni ...).

- Thermal exchange (associated to the configuration of the) (Chen).

- ONB (Onset of Nucleate Boiling) (Hsu and Graham, Bergles and Roshenow, Hsu, Sato and Matsumura, Davis and Anderson, Roshenow and Clark).

- GNV (« Génération nette de vapeur ») (Saha and Zuber, Ünal, Dorra and al, Siman-Tov and al,

- Overheating (McAdams and al, Jens and Lottes, Forster and Greif, Thom and al, Ricque and Siboul).

- Void Rate (mass exchange between phases).

- Speed Gap (Baroczy, Chisholm, HTFS, CISE).

- Critical Flux (Gambill, Katto, Bowring, Sudo and al).

Assessment of the models has shown that:

- We now have models that have been both proven and tested which, furthermore, are potentially applicable to the JHR (corrected by correlation coefficients in order to stick to the measurements that are to be done at the time of the test campaign on the SULTAN-JHR section).

- The case of natural convection still remains and has not yet been dealt with.

- There are fields in which the models are still very imprecise (particularly the critical flux).

Qualification tests :

The first test section is made up of a rectangular channel uniformly heated by two walls.

The geometry of the channel test section is the following:

- Water channel 1.5 mm,
- Width 52 mm (of which 50 mm heating),
- Height 740 mm (70 mm adiabatic + 600 mm heating + 70 mm),
- Inconel wall 600 1 mm. thickness

The operating range of the test loop, SULTAN-JHR is the following:

- Pressure : from 0.2 to 0.9 MPa,
- Inlet temperature : 25 C at Tsat – 5°C,
- Power : from 20 to 600 kW (flux from 0.3 to 10 MW/m^2),
- Flow rate : from 0.05 to 1.4 kg/s (mass flow rate from 650 to 18 000 kg/s/m2).

The following measurements are to be carried out :

Limit conditions :

- flow (2 measurements),
- inlet temperature (2 measurements),
- outlet temperature,
- inlet and outlet pressure,
- thermal flux.

Along the test section :

-wall temperature (mean values and fluctuations)
 . 44 thermal-couples in the heating zone
- detection of the boiling crisis
 . 6 thermo-couples in the adiabatic outlet zone (in order to characterize the recondensation)
- pressure : pressure measurement
 . 1 upstream from the test section
 . 2 in the adiabatic part of the inlet
 . 3 in the heating zone
 . 2 in the adiabatic part of the outlet
 . 1 downstream from the test section.

There will no measurement of the void rate (even though it is a very interesting parameter) as the uncertainties of the measurements of this parameter would be too great to provide anything of real value or interest.

2.4 Studies on the monobloc tank-housing

The housing of the reactor core ensures the major functions of support for the fuel elements as well as those of the water channels, cooling the core. Equipped along its periphery with thermal shields, it limits the heat coming from the core in the experiments located on the periphery.

Given the specific power of 600 KW/l, which was planned for the reference version, it is submitted to a high neutronic flux, with a very high heating (See Figure 1), and at high flowrates (15 m/s). On the other hand, the primary circuit is closed and pressurized at about 5 bar at the outlet of the core and 10 bar at the entrance.

The housing is made up of:

- a massive central part pierced with vertical holes (Figure 2), each holding a fuel element or an experimental device,
- higher and lower parts which are welded onto the preceding part (Figure 3); their main purpose is to convey the water coolant.

The section of the housing can be either rectangular or hexagonal in shape. The neutronic and thermal stresses have led us to select an aluminum alloy for the housing itself and zircaloy for the thermal screens. The mechanical stresses are sustained by the aluminum housing and, in order to demonstrate its feasibility, it was necessary to carry out a number of studies.

2.4.1 Thermal-mechanical calculations for the monobloc tank-housing

The preliminary calculations using the finite elements in 3D on various geometries, when considering a non irradiated material, have shown (see Figures 4 and 5) that:

- the temperatures reached in the core of the alloy remained lower than 80°C,
- the membrane stresses in the isthms could reach values of 60 to 80 Mpa in linkup zones existing between the massive, central part and the higher and lower parts. Therefore, particular attention must be given to the design of these linkups.

Two aluminum alloys are now under consideration:

- the 5754 0 (referred to in France as the AG_3NET) which was used to great extent for all French experimental reactors,
- the 6061 T 6, which is in general use in the United States for experimental reactors.

The composition of these alloys with specific restrictions on the B, Cd, Li et Co are illustrated in Table 6.

These alloys differ in their fabrication mode (the former is annealed and the latter is quenched), which leads to certain differences in the main mechanical characteristics at ambient temperature.

	AG_3NET	6061 T6
Minimal yield strength at temperature	80 MPa	240 MPa
Strength to minimal tension	180 MPa	260 MPa
Sm value	53 MPa	87 MPa
Total elongation at failure	18 %	12 %

Two other differences between these alloys are important for the housing:
- the weldability is much more difficult for the 6061 than it is for the AG_3NET with a significant loss of mechanical characteristics,
- the aging under irradiation is slower for the 6061 than it is for the AG_3NET (problem of the housing lifetime)

Regarding the erosion under irradiation, no data is available in the RJH conditions (15 m/sec).

2.4.2 Mechanical properties of the alloys considered for the housing

Bibliographic research has been carried out on the available knowledge of the characteristics of these two aluminum alloys when they are both new and irradiated. Lack of data in this area has been identified and it would be advisable to launch test programs in order to gain additional information. This should not be a problem for new alloys ; what is required is to provide the matter and carry out the necessary tests. On the other hand, for irradiated alloys, in most cases the matter is lacking or it has been irradiated in quite different flux conditions (spectrum) or the fluences are insufficient.

The major lack of knowledge is summarized in the following two paragraphs :

2.4.2.1 Knowledge of the alloys in non irradiated state

Lack of information concerning the AG_3NET and the 6061 T6.

The variation of the Young modulus beyond ambient temperature.
The evolution of the monotone tension curve beyond ambient temperature.
The evolution of the cyclic curve beyond ambient temperature.
The evolution of the elongations in primary creep.

2.4.2.2 Knowledge of the alloys in an irradiated state

- Lack of knowledge concerning the AG_3NET

The quantified evaluation of the effects of irradiation on the tension characteristics beyond a fluence of 10^{22} n/cm2.
The quantification of irradiation creep.
The strength to brutal failure.

Regarding the first point, tests are now being carried out on the former housing of the OSIRIS reactor, which was replaced after more than 30 years of service.
As for the two other points, complementary tests have been envisioned and they are to be carried out on the same housing or on other available irradiated parts at the CEA.

- Lack of knowledge concerning the 6061 T6:

The quantification of the irradiation creep.
The strength to brutal failure.

Regarding these last two points, it is important to find irradiated matter in foreign reactors.

2.4.2.3 Corrosion/Erosion

For this problem very little information is available. As a result, tests performed in a loop with demineralized water (at a maximum temperature of 50°C) on samples taken from the two alloys are to be carried out. We initially have only planned to perform tests on non-irradiated samples.

2.4.3 Feasibility of the 6061 T6 housing

Given the experience of the French in implementing the AG_3NET, we know that the housing can be manufactured with using this alloy. On the other hand, in the case of the 6061 T6, we are still facing a certain number of technical problems: the feasibility of an adequate and homogeneous quenching, the limitation of the strains at the time of the liberation of the residual stresses during machining and the quality of the welds.

Quenching and limitation of the stresses during machining

To solve these two technical problems, we have planned to produce two “demonstrators”: the first will allow us to define the parameters of the quenching operation and to verify, through destructive tests, that the expected characteristics are obtained; the second test will allow us to perfect the machining method.

Welding

Very little information is available concerning the process of electron welding.

We do not expect to gather all the information we need in bibliographic reviews so that we are now planning qualification tests with measurements of the mechanical characteristics of the welded seams.

2.4.4 The design and construction code:

Apart from the JHR project, the CEA has decided to write a design and construction code derived from the RCCM and the RCC-MR and which will be applied to experimental reactors and devices. This code is called the RCC-MX. It brings together all the general rules of analysis applicable to Level 1 materials, therefore the housing. This document which is now being written, defines according to 3 factors, the design basis with regard to the following types of damage that might be incurred:
Type P damage (monotone).
Type S damage (cyclical)
Buckling
Special rules for the bolting.

2.5 The fuel element design

The design of the JHR fuel element must be the result of an optimization considering not only technical criteria such as the flux levels or the experimental capacity within the core, but also economic criteria such as the cycle duration or the element manufacturing costs.

2.5.1 Areas of optimisating the different parameters

2.5.1.1 Uranium density in the core

This parameter is considered to be essential, along with other factors, in the very determined approach of reducing the operation costs. It can be easily observed that designers in the 1960’s were much more preoccupied with obtaining technical performances than with taking into consideration questions of an economic nature. Taking the operation cost into account when designing the reactor with the purpose of minimizing the operating teams integrated doses is now considered to be of capital importance by the investors.
The implementation of a non-proliferation policy leading to a ban on the use of very enriched uranium in research reactors has had the consequence of increasing the consumption of fuel elements because it has not been possible up until now to compensate for this imposed drop in enrichment by a corresponding uranium density in the elements.

In adopting such an approach in cost reduction, all parameters, which can contribute to increasing the uranium load in the core, must be considered:
- the fuel type (U_3Si_2, $UMo_{5\ \mathrm{to}\ 10}$),
- the volumic fraction,
- the thickness of the fuel meat,
- the spacing between the fuel plates,
- the spacing between the elements in the core.

The fuel type

The U_3Si_2 fuel was firstly considered to be the only fuel that could reach the high uranium densities, particularly with regard to the possibilities of the UA1 fuels.
All the JHR studies were launched on the basis of the U_3Si_2 (4.8 g d’U/cm^3) fuel.
The UMo fuels offer new possibilities. Among the possible candidates (UMo_5, ...UMo_{10}), the choice of the UMo_7 fuel is the result of a compromise between the advantages and drawbacks of a high or low percentage in Mo, considering that the behaviour under irradiation was not fundamentally different between these two extremes. This point was verified in the IRIS irradiation in which a UMo_7 plate and a UMo_9 plate were irradiated simultaneously in the same conditions, revealing no significant difference in swelling according to the burnup.

The volumic fraction

The existing technology allows us to reach high volumic loads in the fuel. Today we consider that the maximum is at 53%.

These high loads have been the object of special manufacturing by the CERCA. Full size plates and full size fuel elements were irradiated in representative conditions. The behavior of the U_3Si_2 fuels under irradiation whose uranium density is beyond 4.8 g U/cm^3 and rising up to 6 g U/cm^3 was considered satisfactory in the test conditions which are the same as those of the OSIRIS and SILOE reactors.

IRRADIATION OF A RANGE OF VOLUMIC FRACTIONS

VOLUMIC FRACTION (%)	DENSITY (G U/CM3)	QUALIFICATION
42.5	4.8 U_3Si_2	Reference fuel Good behavior, max swelling 35 µm
51.8 52.5	8.3 UMo_7 8.1 UMo_9	IRIS, OSIRIS good behavior, max local swelling 28 µm IRIS, OSIRIS good behavior, max local swelling 28 µm
51.3	5.8 U_3Si_2	- IRIS, 4 standard SILOE plates - Burnup : 55 % - max swelling: 22 µm - good behavior
51.3	5.8 U_3Si_2	- SILEX, 2 standard OSIRIS elements - burnup : 74 % - good behavior
53	6.0 U_3Si_2	- IRIS, 4 standard SILOE plates - Burnup : 56 % - max swelling : 29 µm - good behavior

However, the increase of the volumic fraction, in the vicinity of the technological limit, leads to a lowering in the thermal conductibility (smaller proportion of aluminum, porosity increase), which might turn out to problematic.

In the case of very high performance reactors (such as the JHR), the use of high volumic fraction fuels could have consequences on the maximum temperature in the grains of the U_3Si_2 (and their capacity to react with the aluminum)and on the level of thermal strains at the meat/cladding or plate/structure interfaces.

Bearing these considerations in mind, the fuel selected for the feasibility studies phase of the JHR was therefore the U_3Si_2 at 4.8 of U/cm^3 (a volumic fraction of 42.5%).

The volumic fraction for the fuel of the JHR was so limited at a maximum of 50%, in other words, below the technological limit (53%), so as to guarantee the good thermal-mechanical properties of the UMo-Al mixture. This volume fraction leads in the case of the UMo7, the selected alloy, to a density of 8.0 g U/cm^3.

Thickness of the fuel meat

All the fuels having laminated plates of the UAl, U_3 Si_2 type are 1.27 mm thick, i.e. 1/20 of an inch.

This value, which was set at the beginning of the 1960s, is arbitrary and can be slightly increased without challenging the manufacturing process of the laminated fuels or the behaviour under irradiation thereby permitting great benefit to the increase in quantity of fissile materials in the core. The only consequence is an increase in the temperature in the fuel meat, but this is not considered to be critical for the performances, particularly in the case of the UMo fuels.

This possibility was examined in 1997. The manufacturing of plates with thick meat core (0.59 mm instead of 0.51 mm) has not caused any specific problems. The behavior under irradiation of the thick core plates is close to that of the reference core plates.

IRRADIATIONS	FUEL	OBSERVATIONS
IRIS (SILOE)	- 1 standard plate - U_3Si_2 ; 4.8 g U/cm^3 - e = 0.59 mm	Swelling of the plate is negligible at the end of irradiation for a burnup of 50.7%, the maximum swelling is 35 µm.
SILEPE (OSIRIS)	- 1 entire element OSIRIS standard - U3Si2 ; 4.8 g U/cm^3 - e = 0.59 mm	Good behavior of the element which has reached a burnup of 73%

For the Jules Horowitz reactor, it was decided that full advantage must be taken of this possibility in order to increase the uranium load of the core. Nevertheless, plans have been made to remain within the limits, already explored, of about 15 to 20%.

Spacing between the fuel plates

Reducing the space between the plates allows us to “compact” the core, which is a favorable element for the increase in the fissile load and the hardening of the neutrons spectrum. For the Jules Horowitz reactor, one of the objectives is to be able to perform quickly irradiations in the core in order to study damages to the structure materials. So, reducing the width of the water channels remains an objective. This reduction in width, which brings about an increase in the mechanical stresses exerted on the elements, together with more stringent cooling conditions, and which also makes elements manufacturing more complex, must be optimized.

Spacing between the elements in the housing

It was not possible to envision a thickness smaller than 4 mm for considerations concerning the core housing mechanical resistance.

2.6 Looking for the highest performances:

The maximum flow velocity in the fuel elements has been optimized and the inlet temperature in the core has been set at its lowest level, given the geographical situation of the reactor, which is located in CADARACHE.

In such conditions, it is the aluminum cladding temperature that limits the flux performance of the reactor. The temperature limit, which has to be respected in order to avoid the risk of aluminum corrosion, has been set at 150°C following bibliographic review. This limit must be determined precisely from representative experiments.

EXPERIMENTAL RESEARCH PROGRAM TO FIND THE LIMITS

TESTED FUELS	CLADDING TEMPERATURE	DURATION OF THE IRRADIATION BEFORE CLADDING FAILURE	OBSERVATIONS
- UMo_7 : 8.3 g U/cm^3 - UMo_9 : 8.1 g U/cm^3	< 100°C < 100°C	- -	- good behaviour - good behaviour - max local swelling = 28 µm.
- U_3Si_2 : 4.8 g U/cm^3 - U_3Si_2 : 5.8 g U/cm^3 enrichment ^{235}U = 35 % P = 450 – 500 W/cm^2	about 150°C about 150°C	25 Jepp in BR_2	-the visual examination shows some deformation of the fuel plates resulting from high stress levels in the elements. - destructive post irradiation examinations will be carried out at MOL
- UMo_7 : 8.1-8.2 g U/cm^3 - UMo_9 : 8- 8.2 g U/cm^3 enrichment ^{235}U = 35 % and 20 %	about 150°C about 150°C	48 Jepp in PETTEN	- no visual examination yet done. - no means of determining which fuel plate is responsible for the cladding failure - destructive post irradiation examinations are to be carried out at PETTEN.

This program will be pursued and re-oriented according to the findings of the examinations carried out after irradiation. Following these results, the decision was made to consider the use of the AlFeNi as a cladding material rather than the AG3NET. The AlFeNi does indeed show much better mechanical properties and has a better resistance to corrosion; it is presently used in the RHF reactor in GRENOBLE.

CONCLUSION

The Jules Horowitz reactor will replace the OSIRIS reactor in fulfilling the continuous needs for technological irradiations and will allow the study of future reactors requiring high fluxes, especially gas cooled reactors.

The search for high performances, combined with the objective of cost reduction in irradiation experiments has led the CEA to undertake an ambitious R and D program, particularly in the field of the behaviour of materials and fuels under irradiation. This effort has been complemented by test programs aiming at qualifying more accurate neutronic and thermal-hydraulic codes so that the highest performances possible can be reached in terms of flux, while ensuring sufficient safety margins

FIGURE 1
POWER DELIVERED TO THE TANK STRUCTURE

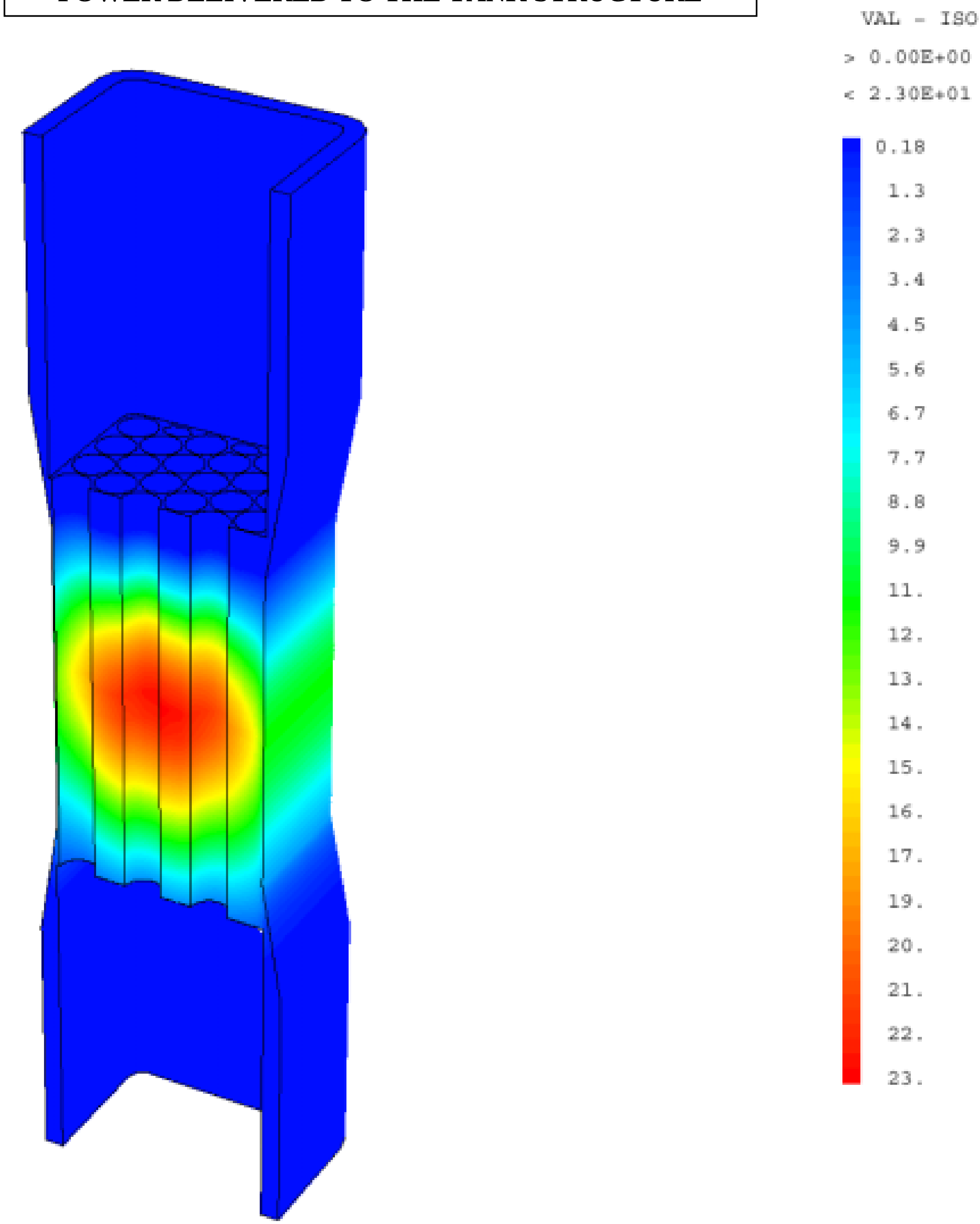


FIGURE 2
CENTRAL PART OF THE TANK

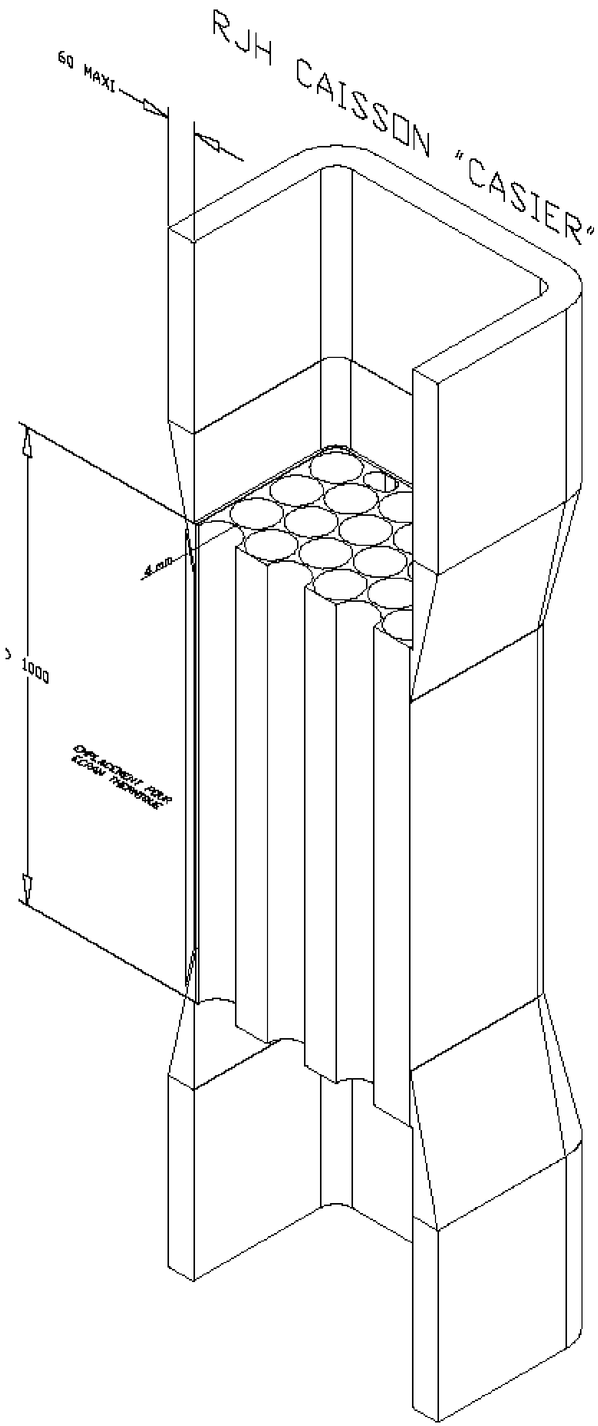
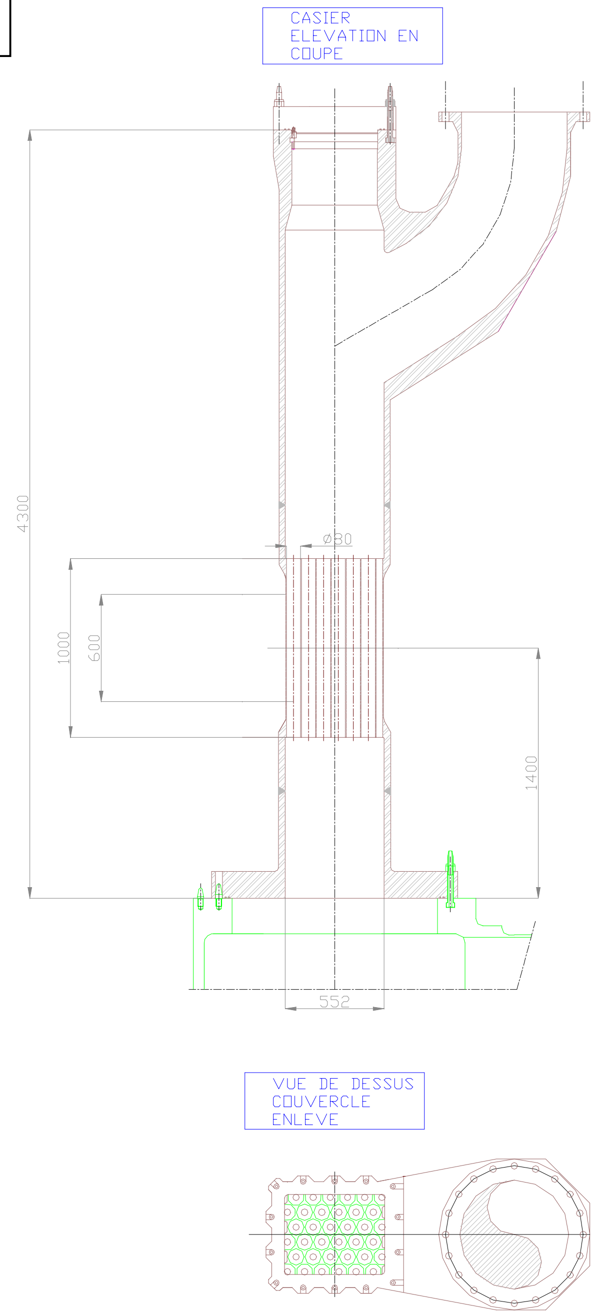


FIGURE 3
REACTOR TANK



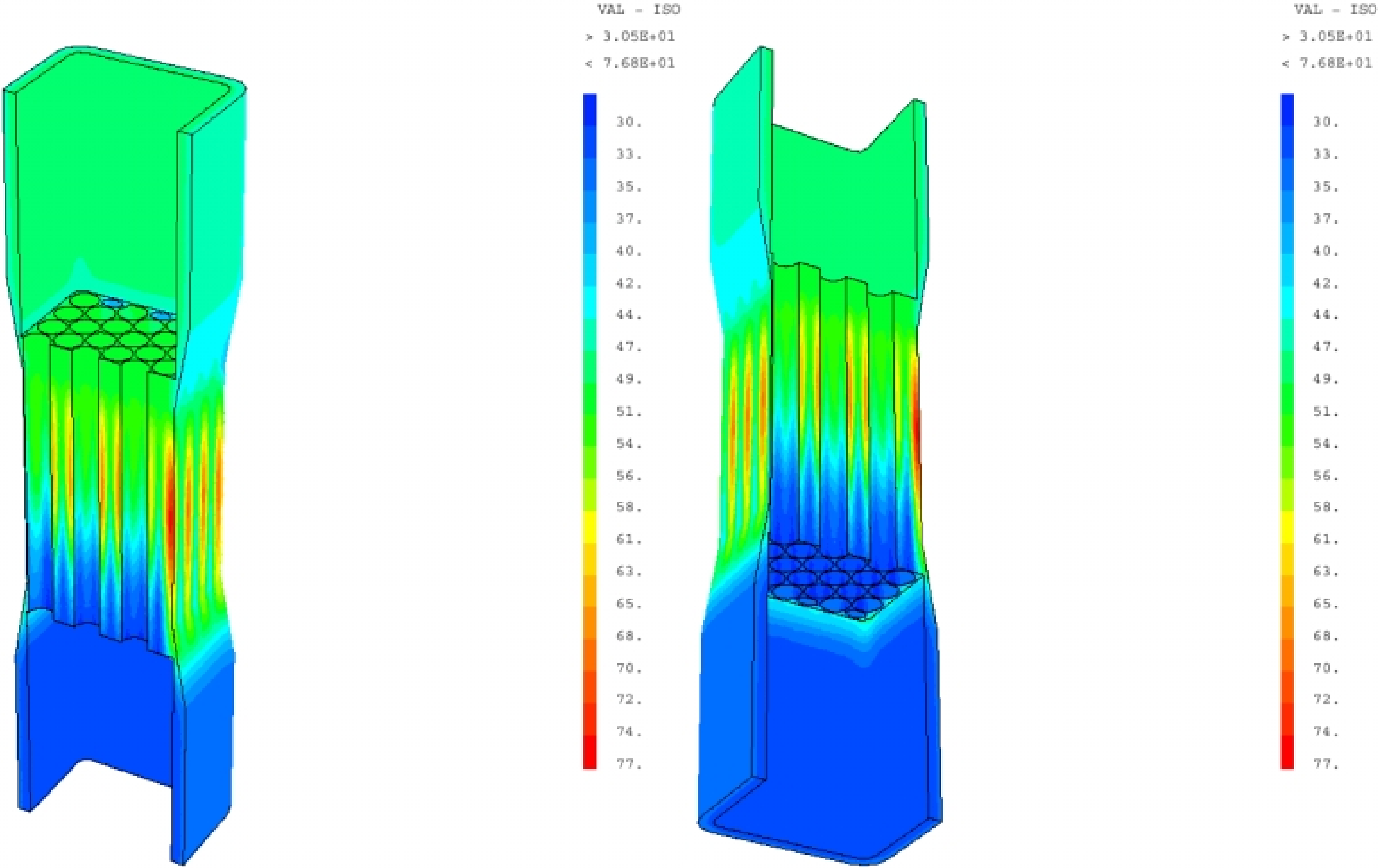


FIGURE 4
TEMPERATURE OF THE TANK STRUCTURE

FIGURE 5
STRESS IN THE TANK STRUCTURE

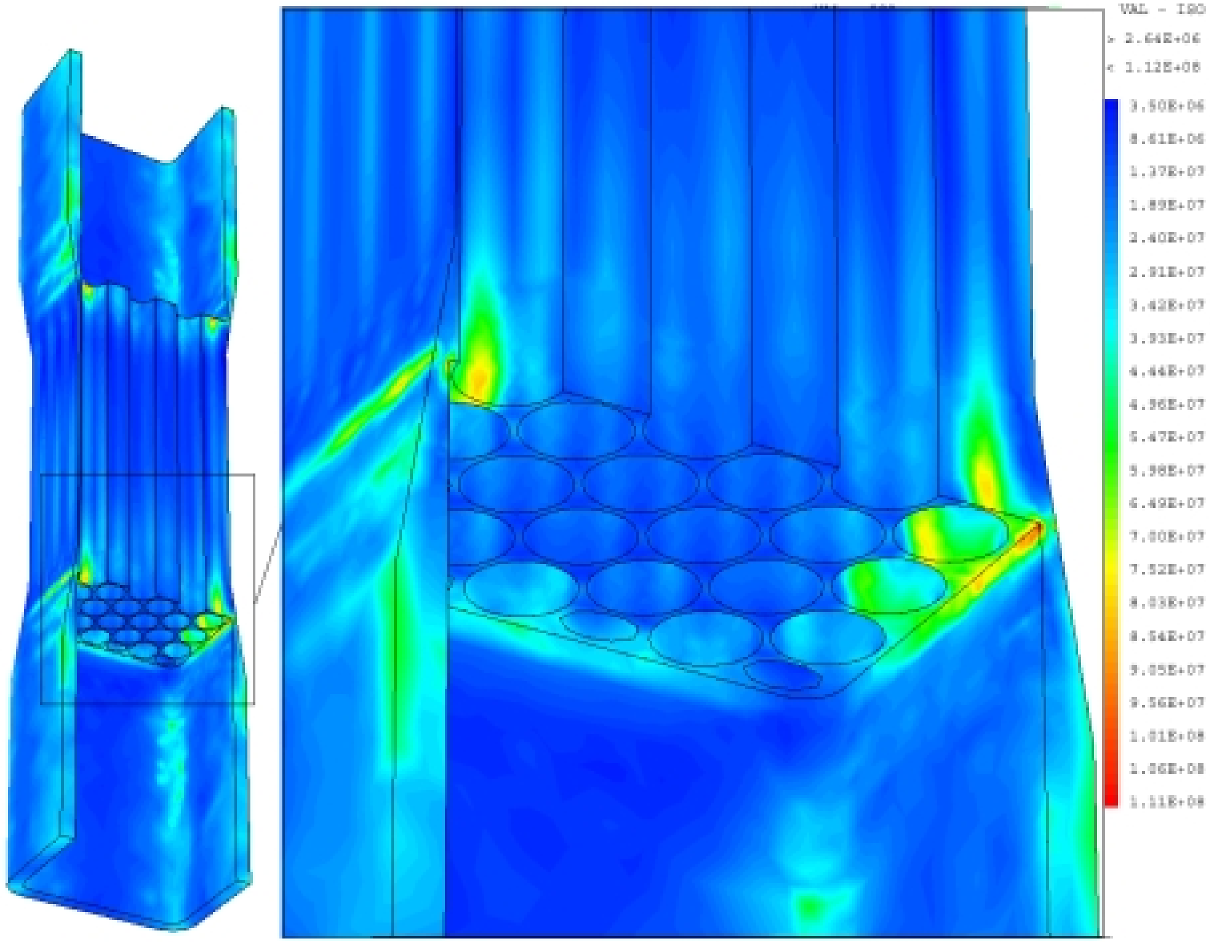


TABLEAU 6
CHEMICAL COMPOSITION

Elements	A5 NET	5754 (standart)	AG3 NET	6061T6 (with added requirements)
Al	base metal	base metal	base metal	base metal
B	≤ 0,001		≤ 0,001	≤ 0,0015
Cd	≤ 0,001		≤ 0,001	≤ 0,001
Co			**≤ 0,000 4**	**≤ 0,001**
Cr		≤ 0,3	≤ 0,3	0,04-0,35
Cu	≤ 0,008	≤ 0,10	≤ 0,008	≤ 0,1
Fe	0,2-0,4	≤ 0,40	0,2-0,4	≤ 0,7
Li	≤ 0,001		≤ 0,001	≤ 0,001
Mg	≤ 0,015	2,6-3,6	2,5-3	0,8-1,2
Mn		≤ 0,50	≤ 0,70	≤ 0,15
Na			≤ 0,001	
Pb			≤ 0,01	
Si	≤ 0,3	≤ 0,40	≤ 0,3	0,4-0,8
Sn			≤ 0,03	
Ti		≤ 0,15	≤ 0,02	≤ 0,15
Zn	≤ 0,03	≤ 0,20	≤ 0,03	≤ 0,08
Fe+Si	0,2-0,5		0,2-0,5	
Mn+Cr		0,1-0,6		
Others (each)	≤ 0,03	≤ 0,05	≤ 0,03	≤ 0,05
Others (total)	≤ 0,6	≤ 0,15	≤ 0,15	≤ 0,15